NON-PUBLIC?: N
ACCESSION #: 9005220384
LICENSEE EVENT REPORT (LER)

FACILITY NAME: COMANCHE PEAK - UNIT 1 PAGE: 1 OF 9

DOCKET NUMBER: 05000445

TITLE: REACTOR TRIP DUE TO ACCIDENTAL BUMPING OF SOURCE RANGE REACTOR
TRIP RESET/BLOCK SWITCH
EVENT DATE: 04/21/90 LER #: 90-009-00 REPORT DATE: 05/18/90

OTHER FACILITIES INVOLVED: N/A DOCKET NO: 05000

OPERATING MODE: 1 POWER LEVEL: 007

THIS REPORT IS SUBMITTED PURSUANT TO THE REQUIREMENTS OF 10 CFR SECTION:
50.73(a)(2)(iv)

LICENSEE CONTACT FOR THIS LER:
NAME: D. NORMAN HOOD SUPERVISOR, COMPLIANCE TELEPHONE: (817) 897-5889

COMPONENT FAILURE DESCRIPTION:
CAUSE: X SYSTEM: SJ COMPONENT: 33 MANUFACTURER: N007
X SJ TGR G080
X SJ CNV F130
X SA BNR X999

REPORTABLE NPRDS: N
N
N
N

SUPPLEMENTAL REPORT EXPECTED: NO

ABSTRACT:

At approximately 1715 CST on April 21, 1990, while dusting the main control board, the Reactor Operator accidentally bumped switch 1/1-N-33B and reset Source Range Channel N31 which had been previously bypassed for power operation. Re-energizing the Source Range high voltage power supply while operating at 7 percent power exceeded the reactor the setpoint and generated a reactor trip signal. The reactor trip

coincident with a low-average Reactor Coolant System (RCS) temperature generated a Feedwater Isolation signal. During reactor trip recovery, the Auxiliary Boiler failed to start due to a faulty igniter assembly and resulted in a slight decrease in RCS Temperature prior to the plant being stabilized in Mode 3.

Corrective actions for this event included the removal of cleaning brushes from the Control Room and the suspension of all control board cleaning until an alternative method is implemented. Additionally, a plexiglass cover has been placed over the involved switch until and evaluation is completed to determine if the switch is overly sensitive and why bumping the switch caused the Source Range Channel to reset.

END OF ABSTRACT

I. DESCRIPTION OF WHAT OCCURRED

A. PLANT OPERATING CONDITIONS BEFORE THE EVENT

On April 21, 1990, Comanche Peak Steam Electric Station (CPSES) Unit 1 was in Mode 1, Power Operation, at approximately 7 percent power. Reactor Coolant System (RCS)(EIIS:(AB)) temperature was at approximately 558 degrees Fahrenheit (F).

B. REPORTABLE EVENT DESCRIPTION (INCLUDING DATES AND APPROXIMATE TIMES OF MAJOR OCCURRENCES)

Event Classification: An event or condition that resulted in an automatic actuation of any Engineered Safety Feature (ESF), including the Reactor Protection System (RPS).

At approximately 1715 CST on April 21, 1990, while dusting the main control board (EIIS:(MCBD)(IB)), the Reactor Operator (utility, licensed) accidentally bumped switch 1/1-N-33B, SR RX TRIP RESET/BLK, (EIIS:(HS)(JG)) with a wooden brush. Bumping the switch resulted in a reactor trip of Unit 1 on Source Range High Flux (SR HF). Upon identification that Unit 1 had tripped, the Reactor Operator and the Relief Reactor Operator (utility, licensed) began performing the immediate actions specified by the Emergency Operating Procedure (EOP) and dispatched Auxiliary Operators (utility, non-licensed) to their watch stations. Motor Driven Auxiliary Feedwater Pumps (EIIS:(P)(BA)) 1-A and 1-B were manually started to feed the

Steam Generators (EIIS:(SG)(AB)) and Steam Generator Blowdown was manually isolated to minimize the cooldown of the RCS as Control Room personnel transitioned from the EOP to the Reactor Trip Response - Emergency Procedure upon the completion of necessary EOP actions.

As would be expected at the 7 percent power level, an ESF actuation occurred (feedwater isolation signal) immediately following the reactor trip due to low-average RCS temperature coincident with a reactor trip. While verifying the expected isolation signal responses
it was identified that position
indication for Steam Generator 1-02 Feedwater Preheater Bypass Valve 1-FV-2194 (EIIS:(FCV)(SJ)) was inconsistent between the handswitch (mid position) and the monitor light box (closed).
In order to provide assurance of the correct valve status, an Auxiliary Operator (utility, non-licensed) was dispatched to visually verify the valve's position.

The Auxiliary Operator checked the position of the limit switches (EIIS:(33)(SJ)) in relation to the valve position and determined that the valve was closed as required. While performing the inspection, the Auxiliary Operator touched the valve's closed limit switch. The cause of the inconsistent position indication was assumed to be a slightly misadjusted/misaligned limit switch since, at the same time the limit switch was touched, the Reactor Operator in the Control Room identified that the handswitch position indication had changed to closed.

The Feedwater Pump Speed Controllers (EIIS:(SCO)(SJ)) were taken to manual and the speed of both Steam Generator Feedwater (SG FW) Pumps (EIIS:(P)(SJ)) was reduced to minimum and the pumps were tripped. SG FW Pump 1-B Recirculation Control Valve 1-FV-2290 (EIIS:(FCV)(SJ)) was closed and an attempt to place the SG FW Pump on its turning gear (EIIS:(TGR)(SJ)) was made when pump speed had been reduced sufficiently. Fuse failure occurred on the SG FW Pump 1-B turning gear due to the turning gear solenoid remaining energized. SG FW Pump 1-B was placed on a "water" turning gear (i.e., condensate flow through the pump) to maintain pump rotation until a fuse was successfully replaced and turning gear operation was established. The Recirculation Control Valve for SG FW Pump 1-A, 1-FV-2289, failed open and would not respond to the main control board controller signal. The immediate cause of the valve failure was determined to be a problem in the valve's air control circuit. The Relief Reactor Operator determined that this was not a significant concern which would affect reactor trip recovery and SG FW Pump 1-A was also placed on a "water" turning gear to maintain pump rotation.

Shortly after the Auxiliary Operators were dispatched, an attempt was made to start the Auxiliary Boiler (EIIS:(BLR)(SA)). However, due to a faulty igniter assembly (EIIS:(BNR)(SA)), the boiler would not start. In order to maintain condenser vacuum without the Auxiliary Boiler, Auxiliary Steam (EIIS:(SA)) was kept lined up to Main Steam (EIIS:(SB)). This resulted in a gradual cooldown of the RCS over approximately 40 minutes which eventually reached a minimum temperature of 545 degrees F. Several actions were taken to minimize primary heat loss, including: turning on the pressurizer heaters (EIIS:(EHTR)(AB)), isolating SG FW Pump steam supplies, transferring from a 75 gallon per minute (gpm) to a 45 gpm letdown orifice (EIIS:(OR)(CB)), and ensuring steam trap drains were re-closed and condenser hot well sparging was secured.

At approximately 1748 CST, the Reactor Trip Response-Emergency Procedure was exited. Subsequently, RCS temperature recovered to approximately 554 degrees F and the steam dumps (EIIS:(PCV)(SB)) were reset. In accordance with approved procedure, plant equipment was secured as necessary and CPSES Unit 1 was stabilized in Mode 3, Hot Standby, at approximately 1820 CST on April 21, 1990.

An event or condition that results in an automatic actuation of any ESF, including the RPS is reportable within 4 hours under 10CFR50.72(b)(2)(ii). At approximately 1815 CST on April 21, 1990 the Nuclear Regulatory Commission Operations Center was notified via the Emergency Notification System of the event as required.

C. STATUS OF STRUCTURES, SYSTEMS, OR COMPONENTS THAT WERE INOPERABLE AT THE START OF THE EVENT AND THAT CONTRIBUTED TO THE EVENT

The igniter assembly for the Auxiliary Boiler was not operating properly at the time of the event and prevented the starting of the Auxiliary Boiler.

D. CAUSE OF EACH COMPONENT OR SYSTEM FAILURE, IF KNOWN, AND CORRECTIVE ACTION TAKEN IN RESPONSE TO EACH FAILURE

1. Several minutes after the reactor trip occurred, position indication for Handswitch 1-HV-2194 was observed by Control Room personnel (utility-licensed) to indicate mid position ('open' and 'closed' lights illuminated) while the monitor light box indicated that Steam Generator 1-02 Feedwater Preheater Bypass Valve 1-FV-2194 was closed (box not illuminated). The cause of the conflicting position indication was due to the closed limit switch on 1-FV-2194 being slightly out of adjustment.

Troubleshooting was subsequently conducted in accordance with a work order, the valve was stroked, and the closed limit switch was adjusted to resolve the position indication concern.

2. The fuse associated with the SG FW Pump 1-B Turning Gear failed several times while attempting to place the pump on the turning gear. The SG FW Pump ring gear failed to engage the clash pinion of the turning gear resulting in the turning gear solenoid remaining energized. Continued energization of the solenoid resulted in the failed fuses. After three attempts, the SG FW Pump was successfully placed on the turning gear.

The engagement problem has occurred before and discussions with the manufacturer-General Electric (GE) have revealed that the engagement problem is typical for this type of turning gear. Previous recommendations provided by GE have been implemented resulting in improved turning gear performance, however, the problem has not been completely resolved. CPSES believes the turning gear supplied by GE may be oversized for its application and a design modification had been initiated prior to the event to install a new design of turning gear.

3. SG FW Pump 1-A Recirculation Control Valve 1-FV-2289 failed too lose upon receipt of a signal from the main control board controller. The cause of this problem was determined to be an open circuit within the valve's

electro-pneumatic converter (EIIS:(CNV)(SJ)). The open circuit was due to 2 of the 4 mounting screws for a coil in the converter vibrating loose. The converter is mounted to the valve bonnet and vibrates with the valve.

The converter was removed and replaced and a new converter was successfully calibrated and returned to service. Subsequently, an evaluation was conducted and the converter will be relocated and mounted on a wall adjacent to the valve. The converter for 1-FV-2290 will also be relocated and mounted away from the valve.

4. Shortly after the reactor trip and the dispatch of the Auxiliary Operators, a start attempt of the Auxiliary Boiler was made and it would not ignite. The inability to start the boiler was due to a crack in the porcelain insulator within the igniter assembly which allowed the ignition spark to occur internal to the igniter assembly rather than at the tip of the igniter. The crack also allowed fuel to migrate into the assembly and foul the igniter. This failure had originally been identified several hours prior to the reactor trip during an attempted startup of the Auxiliary Boiler.

Following the reactor trip, the assembly was cleaned, reinstalled, determined to operate satisfactorily, and utilized until a replacement was procured and installed.

E. FAILURE MODE, MECHANISM, AND EFFECT OF EACH FAILED COMPONENT

See Section 1.B, Reportable Event Description.

F. FOR FAILURES OF COMPONENTS WITH MULTIPLE FUNCTIONS. LIST OF SYSTEMS OR SECONDARY FUNCTIONS THAT WERE ALSO AFFECTED

Not applicable - no failures of components with multiple functions have been identified.

G. FOR FAILURES THAT RENDERED A TRAIN OF A SAFETY SYSTEM INOPERABLE. AN
ESTIMATE OF THE ELAPSED TIME FROM THE DISCOVERY OF INOPERABILITY

UNTIL THE TRAIN WAS RETURNED TO SERVICE

Not applicable - no failures rendering a train of safety system
inoperable have been identified.

H. THE METHOD OF DISCOVERY OF EACH COMPONENT OR SYSTEM
FAILURE OR
PROCEDURAL ERROR

See Section 1.D., Cause of Each Component or System Failure

I. CAUSE OF THE EVENT

Root Cause

The reactor trip occurred as a result of a SR HF Trip Signal
generated due to the accidental bumping of main control board switch
1/1-N-33B. While dusting the main control board around the reactor
controls (EIIS:(JC)(JD)), the Reactor Operator (utility-licensed)
bumped the switch with the wooden portion of a brush and reset
Source Range Channel N31 (EIIS:(CHA)(JC)) which had been previously
bypassed for power operation. Following the detection of a power
level (approximately 7 percent)

in excess of its established reactor trip setpoint (1 E+05 counts
per second), Source Range Channel N31 immediately generated a
reactor the signal. The bumping of the switch is considered a
personnel error Maintenance of main control board cleanliness is not
directly covered by an approved procedure.

Contributory Factor

The Source Range Reactor Trip Reset/Block switch 1/1-N-33B appears
to be overly se
sitive. Although the Reactor Operator accidentally
bumped switch 1/1-N-33B while dusting the main control board, the
impact did not cause the 3-position switch to 1/1-N-33B switch to
rotate sufficiently (compared to what is normally required) to cause
the Source Range Channel to reset.

J. SAFETY SYSTEM RESPONSES THAT OCCURRED

The following safety systems actuated automatically as a result of
the event; the appropriate components within these systems operated

as designed following generation of the SR HF Reactor The Signal and the sensing of Low-Average RCS Temperature coincident with a reactor trip, respectively:

Reactor Protection System (EIIS:(JC))

Feedwater Isolation (EIIS:(SJ))

K. FAILED COMPONENT INFORMATION

1. Limit Switch for Steam Generator 1-02 Feedwater Preheater
Bypass Valve 1 -FV-2194.
Manufacturer: Namco
Model No.: EA18032302

2. Turning Gear for Steam Generator Feedwater Pump 1-B
Manufacturer: General Electric
Model No.: M-567 Lynn Turning Gear

3. Electro - Pneumatic Converter for Steam Generator Feedwater
Pump 1-A
Recirculation Control Valve 1-FV-2289.
Manufacturer: Fisher Controls
Model No.: TYPE 546

4. Igniter Assembly for the Auxiliary Boiler.
Manufacturer: Coen Company
Model No.: 1501 -752-48

II. ASSESSMENT OF THE SAFETY CONSEQUENCES AND IMPLICATIONS OF THIS EVENT

This event consisted of a reactor trip and a feedwater isolation actuation. The reactor the was generated due to a personnel error and was not required by plant conditions. The feedwater isolation was an expected result for a reactor trip at 7 percent power. Both safety systems performed as designed. The event did not result in any challenges to fission product barriers, did not exceed any safety limits, and did not result in any releases of radioactive materials. Therefore, this event did not adversely affect the safe operation of CPSES Unit 1 or the health and safety of the public.

Ill. CORRECTIVE ACTIONS

A. ACTIONS TO PREVENT RECURRENCE

Root Cause
While dusting the main control board around the reactor controls, the Reactor Operator bumped switch 1/1 N-33B with the wooden portion of a brush and reset Source Range Channel N31.

Corrective Action
Brushes utilized for cleaning the main control boards were removed from the control room and a standing order has been issued to the Operations shift crews suspending further cleaning of control boards until an alternative method is implemented.

Contributory Factor
The Source Range Reactor Trip Reset/Block 1/1-N-33B appears to be overly sensitive.

Corrective Action

A Lessons Learned Information Form has been issued to Operations Shift Crews to inform them of the reactor trip event and to caution them of the potential sensitivity of some of the switches on the control board as well as throughout the plant.

At the next appropriate unit shutdown, troubleshooting will be conducted relative to the sensitivity of switch 1/1-N-33B to attempt to determine the reason for causing the inadvertent reset of the Source Range Channel N31.

Plexiglass switch covers have been fabricated and placed over the two Source Range Reactor The Reset/Block Switches and the two Source Range Flux Doubling Reset/Block Switches (EIIS:(HS)(JG)) until the sensitivity concern has been evaluated and addressed.

IV. PREVIOUS SIMILAR EVENTS

There have been no previous similar events reported pursuant to 10CFR50.73.

ATTACHMENT 1 TO 9005220384 PAGE 1 OF 1

Log # TXX-90181
File # 10200
Ref. # 50.73
50.73(a)(2)(iv)

May 18, 1990
W. J. Cahill
Executive Vice President

U. S. Nuclear Regulatory Commission
Attn: Document Control Desk
Washington, D. C. 20555

SUBJECT: COMANCHE PEAK STEAM ELECTRIC STATION
DOCKET NO. 50-445
REACTOR PROTECTION SYSTEM ACTUATION
LICENSEE EVENT REPORT 90-009-00

Gentlemen:

Enclosed is Licensee Event Report 90-009-00 for Comanche Peak Steam Electric Station Unit 1, "Reactor Trip Due to Accidental Bumping of Source Range Reactor Trip Reset/Block Switch."

Sincerely,

William J. Cahill, Jr.

RHS/daj

Enclosure

c - Mr. R. D. Martin, Region IV
Resident Inspectors, CPSES (3)

400 North Olive Street LB 81 Dallas, Texas 75201

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